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NUCLEAR REGULATORY COMMISSION

[NRC-2013-0266]

Biweekly Notice

Applications and Amendments to Facility Operating Licenses and Combined Licenses Involving No Significant Hazards Considerations

Background

Pursuant to Section 189a. (2) of the Atomic Energy Act of 1954, as amended (the Act), the U.S. Nuclear Regulatory Commission (NRC) is publishing this regular biweekly notice. The Act requires the Commission to publish notice of any amendments issued, or proposed to be issued and grants the Commission the authority to issue and make immediately effective any amendment to an operating license or combined license, as applicable, upon a determination by the Commission that such amendment involves no significant hazards consideration, notwithstanding the pendency before the Commission of a request for a hearing from any person.

This biweekly notice includes all notices of amendments issued, or proposed to be issued from November 14, 2013 to November 27, 2013. The last biweekly notice was published on November 26, 2013 (78 FR 70589).

ADDRESSES: You may submit comment by any of the following methods (unless this document describes a different method for submitting comments on a specific subject):

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0266**. Address questions about NRC dockets to Carol Gallagher; telephone: 301-287-3422; e-mail: Carol.Gallagher@nrc.gov. For technical questions, contact the individual(s) listed in the FOR FURTHER INFORMATION CONTACT section of this document.

- **Mail comments to:** Cindy Bladey, Chief, Rules, Announcements, and Directives Branch (RADB), Office of Administration, Mail Stop: 3WFN, 06-44M, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001.

For additional direction on accessing information and submitting comments, see "Accessing Information and Submitting Comments" in the SUPPLEMENTARY INFORMATION section of this document.

SUPPLEMENTARY INFORMATION:

I. Accessing Information and Submitting Comments

A. Accessing Information

Please refer to Docket ID **NRC-2013-0266** when contacting the NRC about the availability of information regarding this document. You may access publicly-available information related to this action by the following methods:

- **Federal Rulemaking Web site:** Go to http://www.regulations.gov and search for Docket ID **NRC-2013-0266**.

- **NRC's Agencywide Documents Access and Management System (ADAMS):** You may access publicly-available documents online in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. To begin the search, select "ADAMS Public Documents" and then select "Begin Web-based ADAMS Search." For problems with ADAMS, please contact the NRC's Public Document Room (PDR) reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov. The ADAMS accession number for each document referenced in this notice (if that document is available in ADAMS) is provided the first time that a document is referenced.

- **NRC's PDR:** You may examine and purchase copies of public documents at the NRC's PDR, Room O1-F21, One White Flint North, 11555 Rockville Pike, Rockville, Maryland 20852.

B. Submitting Comments

Please include Docket ID **NRC-2013-0266** in the subject line of your comment submission, in order to ensure that the NRC is able to make your comment submission available to the public in this docket.

The NRC cautions you not to include identifying or contact information that you do not want to be publicly disclosed in your comment submission. The NRC posts all comment submissions at http://www.regulations.gov as well as entering the comment submissions into ADAMS. The NRC does not routinely edit comment submissions to remove identifying or contact information.

If you are requesting or aggregating comments from other persons for submission to the NRC, then you should inform those persons not to include identifying or contact information that they do not want to be publicly disclosed in their comment submission. Your request should state that the NRC does not routinely edit comment submissions to remove such information before making the comment submissions available to the public or entering the comment submissions into ADAMS.

Notice of Consideration of Issuance of Amendments to Facility Operating Licenses and Combined Licenses, Proposed No Significant Hazards Consideration Determination, and Opportunity for a Hearing

The Commission has made a proposed determination that the following amendment requests involve no significant hazards consideration. Under the Commission's regulations in Section 50.92 of Title 10 of the *Code of Federal Regulations* (10 CFR), this means that operation of the facility in accordance with the proposed amendment would not (1) involve a significant increase in the probability or consequences of an accident previously evaluated; or (2) create the possibility of a new or different kind of accident from any accident previously evaluated; or (3) involve a significant reduction in a margin of safety. The basis for this proposed determination for each amendment request is shown below.

The Commission is seeking public comments on this proposed determination. Any comments received within 30 days after the date of publication of this notice will be considered in making any final determination.

Normally, the Commission will not issue the amendment until the expiration of 60 days after the date of publication of this notice. The Commission may issue the license amendment before expiration of the 60-day period provided that its final determination is that the amendment involves no significant hazards consideration. In addition, the Commission may issue the amendment prior to the expiration of the 30-day comment period should circumstances change during the 30-day comment period such that failure to act in a timely way would result, for example in derating or shutdown of the facility. Should the Commission take action prior to the expiration of either the comment period or the notice period, it will publish in the *Federal Register* a notice of issuance. Should the Commission make a final No Significant Hazards Consideration Determination; any hearing will take place after issuance. The Commission expects that the need to take this action will occur very infrequently.

Within 60 days after the date of publication of this notice, any person(s) whose interest may be affected by this action may file a request for a hearing and a petition to intervene with respect to issuance of the amendment to the subject facility operating license or combined license. Requests for a hearing and a petition for leave to intervene shall be filed in accordance with the Commission's "Agency Rules of Practice and Procedure" in 10 CFR part 2. Interested person(s) should consult a current copy of 10 CFR 2.309, which is available at the NRC's PDR, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. The NRC regulations are accessible electronically from the NRC Library on the NRC's Web site at http://www.nrc.gov/reading-rm/doc-collections/cfr/. If a request for a hearing or petition for leave to intervene is filed by the above date, the Commission or a presiding officer designated by the Commission or by the Chief Administrative Judge of the Atomic Safety and Licensing Board Panel, will rule on the request and/or petition; and the Secretary or the Chief Administrative Judge of the Atomic Safety and Licensing Board will issue a notice of a hearing or an appropriate order.

As required by 10 CFR 2.309, a petition for leave to intervene shall set forth with particularity the interest of the petitioner in the proceeding, and how that interest may be affected by the results of the proceeding. The petition should specifically explain the reasons why intervention should be permitted with particular reference to the following general requirements: 1) the name, address, and telephone number of the requestor or petitioner; 2) the nature of the requestor's/petitioner's right under the Act to be made a party to the proceeding; 3) the nature and extent of the requestor's/petitioner's property, financial, or other interest in the proceeding; and 4) the possible effect of any decision or order which may be entered in the proceeding on the requestor's/petitioner's interest. The petition must also identify the specific contentions which the requestor/petitioner seeks to have litigated at the proceeding.

Each contention must consist of a specific statement of the issue of law or fact to be raised or controverted. In addition, the requestor/petitioner shall provide a brief explanation of the bases for the contention and a concise statement of the alleged facts or expert opinion which support the contention and on which the requestor/petitioner intends to rely in proving the contention at the hearing. The requestor/petitioner must also provide references to those specific sources and documents of which the petitioner is aware and on which the requestor/petitioner intends to rely to establish those facts or expert opinion. The petition must include sufficient information to show that a genuine dispute exists with the applicant on a material issue of law or fact. Contentions shall be limited to matters within the scope of the amendment under consideration. The contention must be one which, if proven, would entitle the requestor/petitioner to relief. A requestor/petitioner who fails to satisfy these requirements with respect to at least one contention will not be permitted to participate as a party.

Those permitted to intervene become parties to the proceeding, subject to any limitations in the order granting leave to intervene, and have the opportunity to participate fully in the conduct of the hearing.

If a hearing is requested, the Commission will make a final determination on the issue of no significant hazards consideration. The final determination will serve to decide when the hearing is held. If the final determination is that the amendment request involves no significant hazards consideration, the Commission may issue the amendment and make it immediately effective, notwithstanding the request for a hearing. Any hearing held would take place after issuance of the amendment. If the final determination is that the amendment request involves a significant hazards consideration, then any hearing held would take place before the issuance of any amendment.

All documents filed in NRC adjudicatory proceedings, including a request for hearing, a petition for leave to intervene, any motion or other document filed in the proceeding prior to the submission of a request for hearing or petition to intervene, and documents filed by interested governmental entities participating under 10 CFR 2.315(c), must be filed in accordance with the NRC's E-Filing rule (72 FR 49139; August 28, 2007). The E-Filing process requires participants to submit and serve all adjudicatory documents over the internet, or in some cases to mail copies on electronic storage media. Participants may not submit paper copies of their filings unless they seek an exemption in accordance with the procedures described below.

To comply with the procedural requirements of E-Filing, at least 10 days prior to the filing deadline, the participant should contact the Office of the Secretary by e-mail at hearing.docket@nrc.gov, or by telephone at 301-415-1677, to request (1) a digital identification (ID) certificate, which allows the participant (or its counsel or representative) to digitally sign documents and access the E-Submittal server for any proceeding in which it is participating; and (2) advise the Secretary that the participant will be submitting a request or petition for hearing (even in instances in which the participant, or its counsel or representative, already holds an NRC-issued digital ID certificate). Based upon this information, the Secretary will establish an electronic docket for the hearing in this proceeding if the Secretary has not already established an electronic docket.

Information about applying for a digital ID certificate is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals/apply-certificates.html. System requirements for accessing the E-Submittal server are detailed in the NRC's "Guidance for Electronic Submission," which is available on the agency's public Web site at http://www.nrc.gov/site-help/e-submittals.html. Participants may attempt to use other software not listed on the Web site, but should note that the NRC's E-Filing system does not support unlisted software, and the NRC Meta System Help Desk will not be able to offer assistance in using unlisted software.

If a participant is electronically submitting a document to the NRC in accordance with the E-Filing rule, the participant must file the document using the NRC's online, Web-based submission form. In order to serve documents through the Electronic Information Exchange System, users will be required to install a Web browser plug-in from the NRC's Web site. Further information on the Web-based submission form, including the installation of the Web browser plug-in, is available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html.

Once a participant has obtained a digital ID certificate and a docket has been created, the participant can then submit a request for hearing or petition for leave to intervene. Submissions should be in Portable Document Format (PDF) in accordance with the NRC's guidance available on the NRC's public Web site at http://www.nrc.gov/site-help/e-submittals.html. A filing is considered complete at the time the documents are submitted through the NRC's E-Filing system. To be timely, an electronic filing must be submitted to the E-Filing system no later than 11:59 p.m. Eastern Time on the due date. Upon receipt of a transmission, the E-Filing system time-stamps the document and sends the submitter an e-mail notice confirming receipt of the document. The E-Filing system also distributes an e-mail notice that provides access to the document to the NRC's Office of the General Counsel and any others who have advised the Office of the Secretary that they wish to participate in the proceeding, so that the filer need not serve the documents on those participants separately. Therefore, applicants and other participants (or their counsel or representative) must apply for and receive a digital ID certificate before a hearing request/petition to intervene is filed so that they can obtain access to the document via the E-Filing system.

A person filing electronically using the agency's adjudicatory E-Filing system may seek assistance by contacting the NRC Meta System Help Desk through the "Contact Us" link located on the NRC's Web site at http://www.nrc.gov/site-help/e-submittals.html, by e-mail to MSHD.Resource@nrc.gov, or by a toll-free call at 1-866 672-7640. The NRC Meta System Help Desk is available between 8 a.m. and 8 p.m., Eastern Time, Monday through Friday, excluding government holidays.

Participants who believe that they have a good cause for not submitting documents electronically must file an exemption request, in accordance with 10 CFR 2.302(g), with their initial paper filing requesting authorization to continue to submit documents in paper format.

Such filings must be submitted by: (1) first class mail addressed to the Office of the Secretary of the Commission, U.S. Nuclear Regulatory Commission, Washington, DC 20555-0001, Attention: Rulemaking and Adjudications Staff; or (2) courier, express mail, or expedited delivery service to the Office of the Secretary, Sixteenth Floor, One White Flint North, 11555 Rockville Pike, Rockville, Maryland, 20852, Attention: Rulemaking and Adjudications Staff. Participants filing a document in this manner are responsible for serving the document on all other participants. Filing is considered complete by first-class mail as of the time of deposit in the mail, or by courier, express mail, or expedited delivery service upon depositing the document with the provider of the service. A presiding officer, having granted an exemption request from using E-Filing, may require a participant or party to use E-Filing if the presiding officer subsequently determines that the reason for granting the exemption from use of E-Filing no longer exists.

Documents submitted in adjudicatory proceedings will appear in the NRC's electronic hearing docket which is available to the public at http://ehd1.nrc.gov/ehd/, unless excluded pursuant to an order of the Commission, or the presiding officer. Participants are requested not to include personal privacy information, such as social security numbers, home addresses, or home phone numbers in their filings, unless an NRC regulation or other law requires submission of such information. However, a request to intervene will require including information on local residence in order to demonstrate a proximity assertion of interest in the proceeding. With respect to copyrighted works, except for limited excerpts that serve the purpose of the adjudicatory filings and would constitute a Fair Use application, participants are requested not to include copyrighted materials in their submission.

Petitions for leave to intervene must be filed no later than 60 days from the date of publication of this notice. Requests for hearing, petitions for leave to intervene, and motions for leave to file new or amended contentions that are filed after the 60-day deadline will not be entertained absent a determination by the presiding officer that the filing demonstrates good cause by satisfying the three factors in 10 CFR 2.309(c)(1)(i)-(iii).

For further details with respect to this license amendment application, see the application for amendment which is available for public inspection at the NRC's PDR, located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through ADAMS in the NRC's Library at http://www.nrc.gov/reading-rm/adams.html. Persons who do not have access to ADAMS or who encounter problems in accessing the documents located in ADAMS should contact the NRC PDR's Reference staff at 1-800-397-4209, 301-415-4737, or by e-mail to pdr.resource@nrc.gov.

Duke Energy Carolinas, LLC, Docket Nos. 50-369 and 50-370, McGuire Nuclear Station, Units 1 and 2, Mecklenburg County, North Carolina

Date of amendment request: September 12, 2013.

Description of amendment request: The proposed amendments revise technical specification 3.3.2, Emergency Safety Feature Actuation System (ESFAS) Instrumentation, to support planned plant modifications associated with NRC Order EA-12-049, Order Modifying Licenses with Regard to Requirements for Mitigation Strategies for Beyond-Design-Basis External Events. Specifically, the amendment modifies the Allowable Value and Nominal Trip Setpoints listed in Table 3.3.2-1, Function 6.f, Auxiliary Feedwater pump suction transfer on low suction pressure.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

Criterion 1:

Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

Response: No.

The proposed TS changes are in support of a plant modification involving the installation of an AC-independent AFW Suction Transfer scheme and hardware to ensure a continuous AFW suction source during an Extended Loss of AC Power (ELAP) event. The purpose of Table 3.3.2-1 Function 6.f is to preserve the AFW pumps by ensuring a continuous suction supply to the pumps. The proposed change will cause the AFW pumps to align to the safety-related suction source sooner than under the current setpoint values for design basis events. The result of the proposed TS setpoint changes will be an increase in margin for AFW pump suction. The new TS setpoints were selected with sufficient margin for instrument uncertainty to ensure that the safety-related AFW suction transfer function actuates before the new AC independent AFW suction transfer function and to prevent any adverse interaction of the two schemes. In other words, the proposed change will ensure the safety-related suction transfer is initiated before the non-safety AC independent AFW suction transfer initiates. The specific TS changes are associated with 1) the specific Nominal Trip Setpoint and Allowable Values for the AFW Pump Suction Transfer on Suction Pressure - Low feature, 2) the addition of specific requirements to be taken if the as-found channel setpoint is outside its predefined as-found tolerance, and 3) the addition of specific requirements regarding resetting of an channel setpoint within an as-left tolerance.

The AFW Pump Suction Transfer on Suction Pressure - Low feature does not affect the probability of any accident being initiated. In addition, none of the abovementioned proposed TS changes affect the probability of any accident being initiated.

Actuation of the AFW Pump Suction Transfer on Suction Pressure - Low feature will continue to ensure that adequate AFW pump suction is maintained during design bases events. Transfer to the safety-related suction source will actually occur earlier due to the proposed change. The proposed changes to Nominal Trip Setpoints and Allowable Values are based on accepted industry standards and will preserve assumptions in the applicable accident analyses. None of the proposed changes alter any assumption previously made in the radiological consequences evaluations, nor do they affect mitigation of the radiological consequences of an accident previously evaluated.

In summary, the proposed changes will not involve any increase in the probability or consequences of an accident previously evaluated.

Criterion 2:

Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

No new accident scenarios, failure mechanisms, or single failures are introduced as a result of any of the proposed changes. The AFW Pump Suction Transfer feature is not an accident initiator. No changes to the overall manner in which the plant is operated are being proposed. Therefore, none of the proposed changes will create the possibility of a new or different kind of accident from any accident previously evaluated.

Criterion 3:

Does the proposed amendment involve a significant reduction in the margin of safety?

Response: No.

Margin of safety is related to the confidence in the ability of the fission product barriers to perform their intended functions. These barriers include the fuel cladding, the reactor coolant system pressure boundary, and the containment barriers. The proposed TS setpoints serve to ensure proper AFW system suction transfer for design bases events, whereby the proposed TS changes will not have any effect on the margin of safety of fission product barriers. In addition, the proposed TS changes will not have any impact on these barriers. No accident mitigating equipment will be adversely impacted as a result of the modification. Therefore, existing safety margins will be preserved. None of the proposed changes will involve a significant reduction in a margin of safety.

Based on the above, it is concluded that the proposed amendment presents no significant hazards consideration under the standards set forth in 10 CFR 50.92 (c), and accordingly, a finding of "no significant hazards consideration" is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lara S. Nichols, Associate General Counsel, Duke Energy Corporation, 526 South Church Street - EC07H, Charlotte, NC 28202.

NRC Branch Chief: Robert J. Pascarelli.

Duke Energy Progress, Inc., Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit 2, Darlington County, South Carolina.

Date of amendment request: September 10, 2013.

Description of amendment request: The proposed change would revise Technical Specification Limiting Condition for Operation 3.8.1, Required Action (RA) B.3.2.2, "One DG [Diesel Generator] Inoperable - Perform SR [Surveillance Requirement] 3.8.1.2 for OPERABLE DG within 96 hours," by a NOTE clarifying RA B.3.2.2 that states, "Not required to be performed when the cause of the inoperable DG is pre-planned maintenance and testing."

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change eliminates a conditional surveillance of the Operable EDG [emergency diesel generator] whenever the alternate division EDG is out of service for pre-planned maintenance and testing. The EDG are [is] not an initiator of any accident previously evaluated. As a result, the probability of any accident previously evaluated is not significantly increased.

   The consequences of any accident previously evaluated are not increased, as the EDG will continue to meet its safety function to supply backup AC [alternating current] power as specified in the accident analysis, in a highly reliable manner, as a common cause problem between the two EDGs will have been precluded, the alternate division EDG will no longer be taken out of service for testing, and its normally scheduled surveillances will be met.

   Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   No new or different accidents result from utilizing the proposed change. The changes do not involve a physical alteration of the plant (i.e., no new or different type of equipment will be installed) or a change in the methods governing normal plant operation. The changes do not alter assumptions made in the safety analysis for EDG performance.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change eliminates a conditional surveillance of the Operable EDG whenever the alternate division EDG is out of service for pre-planned maintenance and testing. The EDG will continue to meet its specified safety function in the safety analysis to provide backup AC power, in a highly reliable manner, as a common cause problem between the two EDGs will have been precluded, the alternate division EDG will no longer be taken out of service for testing, and its normally scheduled surveillances will be met.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Lara S. Nichols, Deputy General Counsel, Duke Energy Corporation, 550 South Tryon Street, Charlotte, NC 28202.

NRC Branch Chief: Jessie F. Quichocho.

Duke Energy Progress, Inc., Docket No. 50-261, H. B. Robinson Steam Electric Plant, Unit 2, Darlington County, South Carolina.

Date of amendment request: September 30, 2013.

Description of amendment request: The proposed amendment implements the Nuclear Regulatory Commission (NRC)-approved Technical Specification Task Force (TSTF) Standard Technical Specification Change Traveler, TSTF-491, "Removal of Main Steam and Main Feedwater Valve Isolation Times from Technical Specifications," via the Consolidated Line Item Improvement Process (CLIIP). This request will modify the current Unit 2 Technical

Specifications (TSs) 3.7.2, Main Steam Isolation Valves and 3.7.3, Main Feedwater Isolation Valves, Main Feedwater Regulation Valves and Bypass Valves by relocating the specific isolation time for the isolation valves from the associated Surveillance Requirements (SRs). The isolation time in the TS SRs is replaced with the requirement to verify the valve isolation time is “within limits.” The specific isolation times will be maintained in the Unit 2 Technical Requirements Manual.

The NRC staff published a notice of opportunity for comment in the *Federal Register* on October 5, 2006 (71 FR 58884), on possible amendments adopting TSTF-491, Revision 2, including a model safety evaluation and model no significant hazards consideration (NSHC) determination, using the CLIIP. The NRC staff subsequently issued a notice of availability of the models for referencing in license amendment applications in the *Federal Register* on December 29, 2006 (71 FR 78472). The licensee affirmed the applicability of the following NSHC determination in its application dated September 30, 2013.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), an analysis of the issue of no significant hazards consideration is presented below:

> Criterion 1:
>
> The Proposed Change Does Not Involve a Significant Increase in the Probability or Consequences of an Accident Previously Evaluated.
>
> The proposed change allows relocating main steam and main feedwater valve isolation times to the Licensee Controlled Document that is referenced in the Bases. The proposed change is described in Technical Specification Task Force (TSTF) Standard TS Change Traveler TSTF-491 related to relocating the main steam and main feedwater valves isolation times to the Licensee Controlled Document that is referenced in the Bases and replacing the isolation time with the phase, ‘‘within limits.’’
>
> The proposed change does not involve a physical alteration of the plant (no new or different type of equipment will be installed). The proposed changes relocate the main steam and main feedwater isolation valve times to the Licensee Controlled Document that is referenced in the Bases. The requirements to perform the testing of these isolation valves are retained in the TS. Future changes to the Bases or licensee-controlled document will be evaluated pursuant to the requirements of 10 CFR 50.59, ‘‘Changes, test and experiments,’’ to

ensure that such changes do not result in more than minimal increase in the probability or consequences of an accident previously evaluated.

The proposed changes do not adversely affect accident initiators or precursors nor alter the design assumptions, conditions, and configuration of the facility or the manner in which the plant is operated and maintained. The proposed changes do not adversely affect the ability of structures, systems and components (SSCs) to perform their intended safety function to mitigate the consequences of an initiating event within the assumed acceptance limits. The proposed changes do not affect the source term, containment isolation, or radiological consequences of any accident previously evaluated. Further, the proposed changes do not increase the types and the amounts of radioactive effluent that may be released, nor significantly increase individual or cumulative occupation/public radiation exposures.

Therefore, the changes do not involve a significant increase in the probability or consequences of any accident previously evaluated.

Criterion 2:

The Proposed Change Does Not Create the Possibility of a New or Different Kind of Accident from any Previously Evaluated.

The proposed changes relocate the main steam and main feedwater valve isolation times to the Licensee Controlled Document that is referenced in the Bases. In addition, the valve isolation times are replaced in the TS with the phase ‘‘within limits.’’ The changes do not involve a physical altering of the plant (i.e., no new or different type of equipment will be installed) or a change in methods governing normal pant operation. The requirements in the TS continue to require testing of the main steam and main feedwater isolation valves to ensure the proper functioning of these isolation valves.

Therefore, the changes do not create the possibility of a new or different kind of accident from any previously evaluated.

Criterion 3:

The Proposed Change Does Not Involve a Significant Reduction in the Margin of Safety.

The proposed changes relocate the main steam and main feedwater valve isolation times to the Licensee Controlled Document that is referenced in the Bases. In addition, the valve isolation times are replaced in the TS with the phase ‘‘within limits.’’ Instituting the proposed changes will continue to ensure the testing of main steam and main feedwater isolation valves. Changes to the Bases or license controlled document are performed in accordance with 10 CFR 50.59. This approach provides an effective level of regulatory control and ensures that main steam and feedwater isolation valve testing is conducted such that there is no significant reduction in the margin of safety.

> The margin of safety provided by the isolation valves is unaffected by the proposed changes since there continue to be TS requirements to ensure the testing of main steam and main feedwater isolation valves. The proposed changes maintain sufficient controls to preserve the current margins of safety.
>
> The NRC staff proposes to determine that the amendment request involves NSHC.

Attorney for licensee: Lara S. Nichols, Deputy General Counsel, Duke Energy Corporation, 550 South Tryon Street, Charlotte, NC 28202.

NRC Branch Chief: Jessie F. Quichocho.

Entergy Gulf States Louisiana, LLC, and Entergy Operations, Inc., Docket No. 50-458, River Bend Station, Unit 1, West Feliciana Parish, Louisiana

Date of amendment request: July 29, 2013.

Description of amendment request: The amendment would add a permanent exception to the River Bend Station (RBS) Technical Requirements Manual (TRM) Section 3.9.14, "Crane Travel - Spent and New Fuel Storage, Transfer, and Upper Containment Fuel Pools," to allow for movement of fuel pool gates over fuel assemblies for maintenance. This exception will also be described by revision to the RBS Updated Safety Analysis Report (USAR) Section 9.1.2.2.2, "Fuel Building Fuel Storage," and Section 9.1.2.3.3, "Protection Features of Spent Fuel Storage Facilities."

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Involved a significant increase in the probability or consequences of an accident previously evaluated.

   Response: No.

   The RBS fuel building fuel storage facilities consist of three interconnected stainless steel-lined concrete pools. The spent fuel storage pool is the largest of these pools. Adjacent to the fuel storage pool are the cask pool and the lower IFTS [inclined fuel transfer system] pool. Each of these two pools is separated from the fuel storage pool by a full-height wall encompassing a watertight gate. The watertight gates are normally open, but are closed to seal their respective pools during cask handling and equipment maintenance operations. It is necessary to lift the gates from the pools for maintenance or seal replacement. The total weight of the gate including the rigging equipment is 2000 pounds. This lift is considered as a heavy load lift since it is higher than the current analyzed light load limit of 1200 pounds for movement of loads over fuel assemblies. TRM 3.9.14 prohibits any load in excess of 1200 pounds from travel over fuel assemblies in the storage pool.

Each of the gates is designed with a pneumatic seal that, when pressurized, seals the respective pool from the spent fuel pool, forming a watertight barrier. No provisions for moving the gates over fuel assemblies were included in the current licensing basis for RBS heavy loads. However, the service life qualification of the gate seals necessitates that they be replaced several times over the life of the plant. Therefore, approval of an exception to the current prohibition is required to allow for replacement of the gate seals.

To perform the movement of the gate from its installed position to a position where the seal can be replaced, an engineering plan that meets the intent of the applicable regulatory guidance has been developed. RBS' program for control of heavy load movements complies with that guidance, and this will prevent the gate from dropping onto the spent fuel assemblies during the movement activity. The program features include the design of the lifting devices, design of the cask and fuel bridge cranes, crane operator training, and the use of written procedures. The regulatory guidance will be met in all respects, except that, in lieu of a single failure-proof crane, the method will employ redundant and diverse means to meet the intent of single-failure proof movements.

Entergy proposes to lift the spent fuel pool gate using a rigging method that complies with the intent of the guidance of References 10.c through 10.f [of the licensee's letter dated July 29, 2013]. The proposed method will be accomplished through the use of fuel building bridge crane and the cask crane at the same time to provide the redundancy required to make the lift single-failure proof and satisfy single-failure proof criteria.

In the proposed method, the fuel building bridge crane and the cask crane will be used to perform the gate lifting and movement. The intent of the applicable regulatory guidance is that in lieu of providing a single-failure-proof crane system, the control of heavy loads guidelines can be satisfied by establishing that the potential for a heavy load drop is extremely small. The gate lifting using the bridge crane and cask crane will conform to applicable regulatory guidelines, in that the probability of the gate drop over the spent fuel assemblies is extremely small. Both cranes have a rated capacity of 15 tons. The maximum weight of the gate and rigging is 2000 pounds. Therefore, there is ample safety factor margin for lifting and movements of the subject spent fuel pool gate. Special lifting devices, which have redundancy or ultimate strength of at least ten times the lifted load, will also be utilized during the rigging process. Even though neither the fuel building bridge crane or the cask crane is a single-failure proof crane, rigging the spent fuel pool gate using both cranes will provide the required redundancy that meets the intent of single-failure proof criteria.

The proposed load lift of the fuel pool gate for replacement of the seal conforms to all of the applicable regulatory guidelines. The design of the lifting lugs and associated rigging (e.g., chains, slings, shackles, hoists, etc.) conforms to the guidelines of NUREG-0612, ["Control of Heavy Loads at Nuclear Power Plants,"] Section 5.1.6, and "Single-Failure Proof Handling System," and References 10.d through 10.f [of the licensee's letter dated July 29, 2013]. The auxiliary hook of the cask crane has a rated capacity of 15 tons. The cask crane is not a single-failure-proof crane. However, it meets NUREG-0612 criteria of Section 5.1.1(6) and is designed for seismic loading. As discussed above, the cask crane, alone, will handle the gate only after the gate is located inside the cask pool where drop of the gate above the spent fuel rack is no longer a concern. The cask pool area has been evaluated for an accidental drop of the spent fuel cask. There is no safety-related equipment inside the cask pool. The analyzed maximum weight of the gate and rigging is 2500 pounds. Therefore, there is ample safety factor margin for lifting the gate with the cask crane.

The probability and consequences of a seismic event are not affected by the proposed gate lift. The consequences of a seismic event during the gate lifting are insignificant since both cranes, the fuel building bridge crane and the cask crane, are seismically qualified for the lifted load. In addition, the design of all rigging conforms to NUREG-0612 guidelines, with a safety factor of 10 for the weight of the load.

Consistent with the defense-in-depth approach outlined in the guidance, the movement will be conducted according to load handling instructions. Operator training will be conducted on the activity prior to the movement, and the equipment will be inspected before the movement will be performed. NUREG-0612 gives guidance that when a particular heavy load must be brought over spent fuel, alternative measures may be used. The combination of preventative measures, as proposed, minimizes the risks inherent in hauling large loads over spent fuel to permissible levels. Considering these provisions and the applicable regulatory guidance, the increase in probability of a load drop is negligible.

It is therefore concluded that the proposed gate lifting and movement does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Create the possibility of a new or different kind of accident from any accident previously evaluated.

   Response: No.

> The lifting of the fuel pool gate in the spent fuel pool as described above minimizes the possibility of a heavy load drop onto spent fuel assemblies as not credible in accordance with single-failure-proof criteria. In addition, movement of the gate in the cask pool using the cask crane does not create the possibility of a new or different kind of accident. The cask drop accident scenario in the current RBS licensing basis (since the cask crane is not a single-failure-proof crane) envelops the accidental drop of the gate in the cask pool during handling by the cask crane. The analyzed weight of a cask is 125 tons, as compared to the 1 ton combined weight of the gate and the rigging.
>
> It is therefore concluded that the proposed gate lifting does not create the possibility of a new or different kind of accident from any previously analyzed.
>
> 3. Invoke a significant reduction in a margin of safety.
>
> Response: No.
>
> By following the guidance of References 10.c through 10.f [of the licensee's letter dated July 29, 2013], the movement of the spent fuel pool gates will have no impact on the analyses of postulated design basis events for RBS. The NRC guidance provides an acceptable means of ensuring the appropriate level of safety and protection against load drop accidents. Therefore, there is no reduction in the margin of safety associated with postulated design basis events at RBS in allowing the proposed change to the RBS licensing basis. RBS will continue to meet its commitment to comply with the applicable guidance.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

<u>Attorney for licensee</u>: Joseph A. Aluise, Associate General Counsel - Nuclear, Entergy Services, Inc., 639 Loyola Avenue, New Orleans, Louisiana 70113.

<u>NRC Branch Chief</u>: Douglas A. Broaddus.

Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois

Date of amendment request: September 5, 2013.

Description of amendment request: The proposed amendments would revise Technical Specification 5.5.13, "Primary Containment Leakage Rate Testing Program," to increase the peak calculated primary containment internal pressure, P_a, from 39.9 psig to 42.6 psig. The proposed increase in P_a reflects a lower initial drywell temperature and a number of other modeling changes.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee provided on September 5, 2013, its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to P_a does not alter the assumed initiators to any analyzed event. The probability of an accident previously evaluated will not be increased by this proposed change since this change does not modify the plant or how it is operated.

   The change in P_a will not affect radiological dose consequence analyses. LSCS radiological dose consequence analyses are based on the maximum allowable containment leakage rate. Even though the test pressure at which leak rate testing is performed is P_a, the maximum allowable containment leakage rate is defined in terms of a percentage of weight of the original content of containment air, which is independent of the peak calculated primary containment internal pressure. The Appendix J containment leak rate testing program will continue to ensure that containment leakage remains within the leakage assumed in the offsite dose consequence analyses. The consequences of an accident previously evaluated will not be increased by this proposed change.

   Therefore, operation of the facility in accordance with the proposed change to Pa will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed change provides a higher P_a than currently described in the TS. This change is the result of a LOCA-Drywell Temperature sensitivity analysis performed by General Electric Hitachi. The peak calculated primary containment internal pressure remains below the containment design pressure of 45 psig. This change does not involve any alteration in the plant configuration (no new or different type of equipment will be installed) or make changes in the methods governing normal plant operation. The change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

Therefore, operation of the facility in accordance with the proposed change to TS 5.5.13 would not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The peak calculated primary containment internal pressure remains below the containment design pressure of 45 psig. LSCS radiological consequence analyses are based on the maximum allowable containment leakage rate. The change in the peak calculated primary containment internal pressure does not represent a significant change in the margin of safety. Operation of the facility in accordance with the proposed change to TS 5.5.13 does not involve a significant reduction in the margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Ms. Tamra Domeyer, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Travis L. Tate.

<u>Exelon Generation Company, LLC, Docket Nos. 50-373 and 50-374, LaSalle County Station, Units 1 and 2, LaSalle County, Illinois</u>

<u>Date of amendment request</u>: September 20, 2013.

<u>Description of amendment request</u>: The proposed amendments would revise Technical Specification 3.3.8.1-1, "Loss of Power Instrumentation," Table 1, to change the allowable values to address non-conservative assumptions. The proposed change involves revising the surveillance requirements to modify the allowable values for the 4.16 kV emergency buses during loss of voltage testing and calibration to ensure that existing design requirements remain satisfied.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee provided on September 20, 2013, its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change to the 4.16 kV [engineered safety functions] ESF bus loss of voltage allowable values allow the protection scheme to function as originally designed. (This change will involve alteration of nominal trip setpoints in the field and will also be reflected in revisions to the calibration procedures.) The proposed change does not affect the probability or consequences of any accident. Analysis was conducted and demonstrates that the proposed allowable values will allow the normally operating safety-related motors to continue to operate without sustaining damage or tripping during the worst-case, non-accident degraded voltage condition for the maximum possible time-delay of 5.7 minutes. Thus, these safety-related loads will be available to perform their safety function if a loss-of-coolant accident (LOCA) concurrent with a loss-of-offsite power (LOOP) occurs following the degraded voltage condition.

   The proposed changes do not adversely affect accident initiators or precursors, and do not alter the design assumptions, conditions, or configuration or the plant or the manner in which the plant is operated or maintained. The proposed allowable values ensure that the 4.16 kV distribution system remains connected to the offsite power system when adequate offsite voltage is available and motor starting transients are considered. The diesel start due to a LOCA signal is not adversely affected by this change. During an actual loss of voltage condition, the loss of voltage time delay will continue to isolate the 4.16 kV distribution system from offsite power before the diesel is ready to assume the emergency loads, which is the limiting time basis for mitigating system responses to the accident. For this reason, the existing loss of power/LOCA analysis continues to be valid.

Therefore, the proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change involves the revision of 4.16 kV ESF bus loss of voltage allowable values to satisfy existing design requirements. The proposed change does not introduce any changes or mechanisms that create the possibility of a new or different kind of accident. The proposed change does not install any new or different type of equipment, and installed equipment is not being operated in a new or different manner. No new effects on existing equipment are created nor are any new malfunctions introduced.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   The proposed protection voltage allowable values are low enough to prevent inadvertent power supply transfer, but high enough to ensure that sufficient power is available to the required equipment. The diesel start due to a LOCA signal is not adversely affected by this change. During an actual loss of voltage condition, the loss of voltage time delays will continue to isolate the 4.16 kV distribution system from offsite power before the diesel is ready to assume the emergency loads.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Ms. Tamra Domeyer, Associate General Counsel, Exelon Generation Company, LLC, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Travis L. Tate.

Exelon Generation Company (EGC), LLC, Docket Nos. STN 50-456 and STN 50-457, Braidwood Station, Units 1 and 2, Will County, Illinois

Date of amendment request: October 10, 2013.

Description of amendment request: The proposed amendment would revise the date for the performance of the containment leakage rate Type A test from “no later than May 4, 2014,” to “prior to entering MODE 4 at the start of Cycle 18.” Additionally, EGC is proposing to establish a requirement for Braidwood Station, Unit 2, to exit the MODEs of applicability for Containment as described in Technical Specification 3.6.1, “Containment” (i.e., MODEs 1 - 4), no later than May 4, 2014.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

> EGC has evaluated the proposed change for Braidwood Station, Units 1 and 2 using the criteria in 10 CFR 50.92, and has determined that the proposed change does not involve a significant hazards consideration. The following information is provided to support a finding of no significant hazards consideration.
>
> 1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?
>
>   Response: No.
>
>   The proposed change to the Braidwood Station, Units 1 and 2 Containment Leakage Rate Testing Program does not involve a physical change to the plant or a change in the manner in which the plant is

operated or controlled. The containment function is to provide an essentially leak tight barrier against the uncontrolled release of radioactivity to the environment for postulated accidents. As such, the containment itself, and the testing requirements to periodically demonstrate the integrity of the containment, exist to ensure the plant's ability to mitigate the consequences of an accident do not involve any accident precursors or initiators. Therefore, the probability of occurrence of an accident previously evaluated is not significantly increased by the proposed amendment. Implementation of the proposed change will continue to provide adequate assurance that during design basis accidents, the containment and its components would limit leakage rates to less than the values assumed in the plant safety analyses. Therefore, the consequences of an accident previously evaluated will not be increased by this proposed change.

Therefore, operation of the facility in accordance with the proposed administrative change to the date for the performance of the Unit 2, Type A containment leak rate test will not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The containment, and the testing requirements to periodically demonstrate the integrity of the containment, exist to ensure the plant's ability to mitigate the consequences of an accident, and do not involve any accident precursors or initiators. The proposed change does not involve a physical change to the plant (i.e., no new or different type of equipment will be installed) or a change to the manner in which the plant is currently operated or controlled.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

   Response: No.

   This proposed change does not alter the manner in which safety limits, limiting safety system setpoints, or limiting conditions for operation are determined. The specific requirements and conditions of the containment leakage rate testing program, as proposed, will continue to ensure that the degree of containment structural integrity and leak-tightness that is considered in the plant's safety analysis is maintained.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

> Based on the above evaluation, EGC concludes that the proposed amendment does not involve a significant hazards consideration under the standards set forth in 10 CFR 50.92, paragraph (c), and accordingly, a finding of no significant hazards consideration is justified.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the requested amendments involve no significant hazards consideration.

Attorney for licensee: Mr. Bradley J. Fewell, Associate General Counsel, Exelon Nuclear, 4300 Winfield Road, Warrenville, IL 60555.

NRC Branch Chief: Travis L. Tate.

PPL Susquehanna, LLC, Docket Nos. 50-387 and 50-388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: June 6, 2013.

Description of amendment request: The proposed amendment would change the current requirement that "each ADS [Automatic Depressurization System] valve opens when manually actuated," to the requirement that "each ADS valve actuator strokes when manually actuated." Additionally, the surveillance frequency would change from "24 months on a STAGGERED TEST BASIS for each valve solenoid," to "24 months."

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The proposed change does not modify the method of demonstrating the operability of the Safety/ Relief Valves (S/RVs) in both the safety and relief modes of operation. The proposed change does modify the method for demonstrating the proper mechanical functioning of the S/RVs. The S/RVs are required to function in the safety mode to prevent overpressurization of the reactor vessel and reactor coolant system pressure boundary during various analyzed transients, including Main Steam Isolation Valve closure. S/RVs associated with the Automatic Depressurization System are also required to function in the relief mode to reduce reactor pressure to permit injection by low pressure Emergency Core Cooling System (ECCS) pumps during certain reactor coolant pipe break accidents. The current testing method demonstrates the proper mechanical functioning of the S/RVs in both modes through manual actuation of the S/RVs. The proposed testing method results in acceptable demonstration of the S/RV functions in both the safety and relief modes, and therefore provides assurance that the probability of S/RV failure will not increase. None of the accident safety analyses are affected by the requested [Technical Specification] TS changes and the consequences of accidents mitigated by the S/RVs will not increase.

Therefore, the proposed amendment does not result in a significant increase in the probability or consequences of any previously evaluated accident.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change modifies the method of testing of the S/RVs, but does not alter the functions or functional capabilities of the S/RVs. Testing under the proposed method is performed in offsite test facilities and in the plant during outage periods when the S/RV functions are not required. Existing analyses address events involving an S/RV inadvertently opening or failing to reclose. Analyses also address the failure of one or more S/RVs to open. The proposed change does not introduce any new failure mode.

   Therefore, it does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

   Response: No.

   The proposed amendment provides for a complete verification of the functional capability of the S/RVs by performing tests, inspections, and maintenance activities without opening the valves while installed in the plant. This alternative testing and associated programmatic controls will provide an overall level of assurance that the S/RVs are capable of

> performing their intended accident mitigation safety functions. The proposed amendment does not affect the valve setpoints or adversely affect any other operational criteria assumed for accident mitigation. No changes are proposed that alter the setpoints at which protective actions are initiated, and there is no change to the operability requirements for equipment assumed to operate for accident mitigation. Moreover, it is expected that the alternative testing methodology will increase the margin of safety by reducing the potential for S/RV leakage resulting from testing. Additionally, the increased testing frequency of the manual actuation circuitry is beneficial since the valves will no longer be tested on a staggered test frequency.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101-1179.

Acting NRC Branch Chief: John G. Lamb.

PPL Susquehanna, LLC, Docket Nos. 50-387 and 50-388, Susquehanna Steam Electric Station, Units 1 and 2, Luzerne County, Pennsylvania

Date of amendment request: June 6, 2013.

Description of amendment request: This proposed change adds a footnote to Function 6c in Technical Specification Table 3.3.6.1-1. This change allows only one Trip System to be operable in MODES 4 and 5 for the Manual Initiation Function for Shutdown Cooling System isolation.

<u>Basis for proposed no significant hazards consideration determination</u>: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Do the proposed changes involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The manual isolation function of the RHR [Residual Heat Removal] Shutdown Cooling System is not credited in any FSAR [Final Safety Analysis Report] safety analysis. The addition of Footnote (c) to the manual isolation function in TS [Technical Specification] Table 3.3.6.1-1 allows one of the two trip systems to be inoperable in MODES 4 and 5 and does not alter any equipment.

   Therefore, this proposed change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Do the proposed changes create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The addition of Footnote (c) to the manual isolation function in TS Table 3.3.6.1-1 allows one of the two trip systems to be inoperable in MODES 4 and 5 and is consistent with other isolation function required for isolation in MODES 4 and 5.

   No new equipment is being introduced, and installed equipment is not being operated in a new or different manner. There are no set points, at which protective or mitigative actions are initiated, affected by this change. These changes do not alter the manner in which equipment operation is initiated, nor will the function demands on credited equipment be changed. No alterations in the procedures that ensure the plant remains within analyzed limits are being proposed, and no major changes are being made to the procedures relied upon to respond to an off-normal event as described in the FSAR. As such, no new failure modes are being introduced. The proposed change does not alter assumptions made in the safety analysis and licensing basis since the manual isolation function of the RHR Shutdown Cooling System is not credited in any FSAR safety analysis.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any previously evaluated.

3. Do the proposed changes involve a significant reduction in a margin of safety?

> Response: No.
>
> The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes are acceptable since no automatic isolation functions are being changed. Since the manual isolation function of the RHR Shutdown Cooling System is not credited in any FSAR safety analysis, this change does not affect the margin of safety assumed by the safety analysis.
>
> Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Bryan A. Snapp, Esquire, Assoc. General Counsel, PPL Services Corporation, 2 North Ninth St., GENTW3, Allentown, PA 18101-1179

Acting NRC Branch Chief: John G. Lamb.

Tennessee Valley Authority, Docket Nos. 50-327 and 50-328, Sequoyah Nuclear Plant, Units 1 and 2, Hamilton County, Tennessee

Date of amendment request: October 2, 2013 (TS-SQN-13-01 and 13-02).

Description of amendment request: The proposed amendments would revise Units 1 and 2 Technical Specifications (TSs) 3.7.5, "Ultimate Heat Sink," to place additional limitations on the maximum average Essential Raw Cooling Water (ERCW) System supply header water temperature during operation with one ERCW pump per loop and operation with one ERCW supply strainer per loop. In addition, the one-time limitations on Unit 1 ultimate heat sink (UHS) temperature and the associated license condition requirements used for the Unit 2 steam generator replacement project are proposed to be deleted. The proposed changes would place additional temperature limitations on the UHS TS Limiting Condition for Operation 3.7.5 with associated required actions, to support maintenance on plant component without requiring a dual unit shutdown.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration determination, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?

   Response: No.

   The proposed change to impose additional limits on UHS temperature while in certain ERCW system alignments does not result in any physical changes to plant safety-related structures, systems, or components (SSCs). The UHS and associated ERCW system function is to remove plant system heat loads during normal and accident conditions. As such, the UHS and ERCW system are not accident initiators, but instead perform accident mitigation functions by serving as the heat sink for safety-related equipment to ensure the conditions and assumptions credited in the accident analyses are preserved. During operation under the proposed change with only one ERCW pump operable in a loop a single failure could cause a total loss of ERCW flow in one loop whereas with two pumps per loop operable only a reduction in flow would occur. In either case, one pump or two pumps per loop operable, the other ERCW loop will continue to perform the design function of the ERCW system. Therefore, the proposed change does not involve a significant increase in the probability of an accident previously evaluated.

   The purpose of this change is to modify the UHS TS to be consistent with the conditions and assumptions of the current design basis heat transfer and flow modeling analyses for the UHS and ERCW system. The proposed change provides assurance that the minimum conditions necessary for the UHS and ERCW system to perform their heat removal safety function is maintained. Accordingly, as demonstrated by TVA design heat transfer and flow modeling calculations, the proposed new requirements will provide the necessary assurance that fuel cladding, Reactor Coolant System (RCS) pressure boundary, and containment integrity limits are not challenged during worst-case post-accident conditions. Accordingly, the conclusions of the accident analyses will remain as previously evaluated such that there will be no significant increase in the post-accident dose consequences.

   Therefore, the proposed change does not involve a significant increase in the probability or consequence of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change does not involve any physical changes to plant safety related SSCs or alter the modes of plant operation in a manner that is outside the bounds of the current UHS and ERCW system design heat transfer and flow modeling analyses. The proposed additional limits on UHS temperature for the specified ERCW system alignments provide assurance that the conditions and assumptions credited in the accident analyses are preserved. Thus, although the specified ERCW system alignments result in reduced heat transfer flow capability, the plant's overall ability to reject heat to the UHS during normal operation, normal shutdown, and hypothetical worst-case accident conditions will not be significantly affected by this proposed change. Since the safety and design requirements continue to be met and the integrity of the RCS pressure boundary is not challenged, no new credible failure mechanisms, malfunctions, or accident initiators are created, and there will be no effect on the accident mitigating systems in a manner that would significantly degrade the plant's response to an accident.

   Therefore, the proposed change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed change modifies the UHS TS to maintain the UHS temperature and associated ERCW system flows within the bounds of the conditions and assumptions credited in the accident analyses. As demonstrated by TVA design basis heat transfer and flow modeling calculations, the additional limits on UHS temperature for the specified ERCW system alignments will provide assurance that the design limits for fuel cladding, RCS pressure boundary, and containment integrity are not exceeded under both normal and post-accident conditions. As required, these calculations include evaluation of the worst-case combination of meteorology and operational parameters, and establish adequate margins to account for measurement and instrument uncertainties. While operating margins have been reduced by the proposed change in order to support necessary maintenance activities, the current limiting design basis accidents remain applicable and the analyses conclusions remain bounding such that the accident safety margins are maintained. Accordingly, the proposed change will not significantly degrade the margin of safety of any SSCs that rely on the UHS and ERCW system for heat removal to perform their safety related functions.

   Therefore, the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Jessie F. Quichocho.

Tennessee Valley Authority, Docket No. 50-390, Watts Bar Nuclear Plant (WBN), Unit 1, Rhea County, Tennessee

Date of amendment request: July 30, 2013.

Description of amendment request: The proposed amendment would modify Technical Specification (TS) 4.3.1.1, "Criticality," to clarify the requirements for storage of new and spent fuel assemblies in the spent fuel racks. This change is necessary to update the current WBN Unit 1 TS to ensure consistency with the proposed TS 4.3.1.1 for WBN Unit 2. In addition, editorial changes are being made to TS 4.3.1. The proposed changes also modify the current licensing basis, as described in Section 4.3.2.7 of the Updated Final Safety Analysis Report (UFSAR).

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration. The NRC staff has reviewed the licensee's analysis against the standards of 10 CFR 50.92(c). The NRC staff's review is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?

   Response: No.

The proposed amendment directs the operators to directly use an existing control figure in the TS instead of a conflicting wording of slightly lower fuel storage enrichment limit in the same section of the TS. No change is being made to the parameters or methodology in evaluated accidents. As a result, there is no increase in the likelihood of existing event initiators.

This figure was supported by the original analyses that determines the subcriticality available in the spent fuel pool and the associated acceptable cell loading patterns have not been changed. Thus the acceptance criteria as stated in the UFSAR are met. Implementing the change involves no facility equipment, procedure, or process changes that could affect the radioactive material actually released during an event. As a result, no conditions have been created that could significantly increase the consequences of any of the events evaluated in the UFSAR.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed changes do not require any new or different accidents to be postulated because no changes are being made to the plant that would introduce any new accident causal mechanism. This license amendment request does not affect any plant systems that are potential accident initiators. The change in TS wording is consistent with an existing figure in the same section of the TS that is bounded by the original plant spent fuel pool criticality analysis. No change to the fuel, spent fuel racks, or spent fuel pool water chemistry are associated with this change.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

   Response: No.

   The proposed amendment directs the operators to directly use an existing control figure in the TS instead of a conflicting wording of slightly lower fuel storage enrichment limit in the same section of the TS. The change in TS wording is consistent with an existing figure in the same section of the TS which is bounded the original plant spent fuel pool criticality analysis. The proposed changes do not alter the permanent plant design, including instrument set points.

Based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Jessie F. Quichocho.

Tennessee Valley Authority, Docket No. 50-390, Watts Bar Nuclear Plant (WBN), Unit 1, Rhea County, Tennessee

Date of amendment request: August 28, 2013.

Description of amendment request: The proposed changes would modify WBN, Unit 1 Technical Specifications (TS) requirements related to direct current (DC) electrical systems. In addition, a new “Battery Monitoring and Maintenance Program” is being proposed. The proposed TS changes place requirements on the battery itself rather than the battery cells as currently required.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed amendment involve a significant increase in the probability or consequence of an accident previously evaluated?

   Response: No.

   The proposed changes restructure the Technical Specifications (TS) for the direct current (DC) electrical power system and are consistent with Technical Specifications Task Force (TSTF) change TSTF-360, Revision 1 and TSTF-500, Revision 2. The proposed changes modify TS Actions relating to battery and battery charger inoperability. The DC electrical power system, including associated battery chargers, is not an initiator of any accident sequence analyzed in the Updated Final Safety Analysis Report (UFSAR). Rather, the DC electrical power system supports equipment used to mitigate accidents. The proposed changes to restructure TS and change surveillances for batteries and chargers to incorporate the updates included in TSTF-360, Revision 1 as updated by TSTF-500, Revision 2, will maintain the same level of equipment performance required for mitigating accidents assumed in the UFSAR. Operation in accordance with the proposed TS would ensure that the DC electrical power system is capable of performing its specified safety function as described in the UFSAR. Therefore, the mitigating functions supported by the DC electrical power system will continue to provide the protection assumed by the analysis. The relocation of preventive maintenance surveillances, and certain operating limits and actions, to a licensee controlled Battery Monitoring and Maintenance Program will not challenge the ability of the DC electrical power system to perform its design function. Appropriate monitoring and maintenance that are consistent with industry standards will continue to be performed. In addition, the DC electrical power system is within the scope of 10 CFR 50.65, "Requirements for monitoring the effectiveness of maintenance at nuclear power plants," which will ensure the control of maintenance activities associated with the DC electrical power system.

The integrity of fission product barriers, plant configuration, and operating procedures as described in the UFSAR will not be affected by the proposed changes. Therefore, the consequences of previously analyzed accidents will not increase by implementing these changes.

Therefore, the proposed amendment does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed amendment create the possibility of a new or different kind of accident from any accident previously evaluated?

Response: No.

The proposed changes involve restructuring the TS for the DC electrical power system. The DC electrical power system, including associated battery chargers, is not an initiator to any accident sequence analyzed in the UFSAR. Rather, the DC electrical power system supports equipment used to mitigate accidents. The proposed changes to restructure the TS and change surveillances for batteries and chargers to incorporate the updates included in TSTF-360 Revision 1 as updated by TSTF-500, Revision 2, will maintain the same level of equipment performance required for mitigating accidents assumed in the UFSAR. Administrative and mechanical controls are in place to ensure the design and operation of the DC systems continues to meet the plant design basis describe in the UFSAR.

Therefore, the proposed amendment will not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed amendment involve a significant reduction in a margin of safety?

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The equipment margins will be maintained in accordance with the plant-specific design bases as a result of the proposed changes. The proposed changes will not adversely affect operation of plant equipment. These changes will not result in a change to the setpoints at which protective actions are initiated. Sufficient DC capacity to support operation of mitigation equipment is ensured. The changes associated with the new battery Maintenance and Monitoring Program will ensure that the station batteries are maintained in a highly reliable manner. The equipment fed by the DC electrical sources will continue to provide adequate power to safety-related loads in accordance with analysis assumptions. TS changes made to be consistent with the changes in TSTF-360, Revision 1, as updated by TSTF-500, Revision 2, maintain the same level of equipment performance stated in the UFSAR and the current TSs.

Therefore, the proposed changes do not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: General Counsel, Tennessee Valley Authority, 400 West Summit Hill Drive, ET 11A, Knoxville, Tennessee 37902.

NRC Branch Chief: Jessie F. Quichocho.

Wolf Creek Nuclear Operating Corporation, Docket No. 50-482, Wolf Creek Generating Station, Coffey County, Kansas

Date of amendment request: September 23, 2013.

Description of amendment request: The amendment would revise Technical Specification (TS) 5.6.5, "CORE OPERATING LIMITS REPORT (COLR)," to replace WCAP-11596-P-A, “Qualification of the Phoenix-P/ANC Nuclear Design System for Pressurized Water Reactor Cores,” with WCAP-16045-P-A, “Qualification of the Two-Dimensional Transport Code PARAGON,” and WCAP-16045-P-A, Addendum 1-A, “Qualification of the NEXUS Nuclear Data Methodology,” to determine core operating limits.

Basis for proposed no significant hazards consideration determination: As required by 10 CFR 50.91(a), the licensee has provided its analysis of the issue of no significant hazards consideration, which is presented below:

1. Does the proposed change involve a significant increase in the probability or consequences of an accident previously evaluated?

   Response: No.

   The analytical methodologies, which this license amendment proposes for determination of core operating limits, are improvements over the current methodologies in use at WCGS. The NRC staff reviewed and approved these methodologies and concluded that these analytical methods are acceptable as a replacement for the current analytical method. Thus core operating limits determined using the proposed analytical methods continue to assure that the reactor operates safely and, thus, the proposed changes do not involve an increase in the probability of an accident.

   Operation of the reactor with core operating limits determined by use of the proposed analytical methods does not increase the reactor power level, does not increase the core fission product inventory, and does not change any transport assumptions. Therefore the proposed methodology and TS changes do not involve a significant increase in the consequences of an accident.

   Therefore, it is concluded that this change does not involve a significant increase in the probability or consequences of an accident previously evaluated.

2. Does the proposed change create the possibility of a new or different kind of accident from any accident previously evaluated?

   Response: No.

   The proposed change provides revised analytical methods for determining core operating limits, and does not change any system functions or maintenance activities. The change does not involve physical alteration of the plant, that is, no new or different type of equipment will be installed. The change does not alter assumptions made in the safety analyses but ensure that the core will operate within safe limits. This change does not create new failure modes or mechanisms that are not identifiable during testing, and no new accident precursors are generated.

Therefore, it is concluded that this change does not create the possibility of a new or different kind of accident from any accident previously evaluated.

3. Does the proposed change involve a significant reduction in a margin of safety?

Response: No.

The margin of safety is established through equipment design, operating parameters, and the setpoints at which automatic actions are initiated. The proposed changes do not physically alter safety-related systems, nor does it affect the way in which safety related systems perform their functions. The setpoints at which protective actions are initiated are not altered by the proposed changes. Therefore, sufficient equipment remains available to actuate upon demand for the purpose of mitigating an analyzed event. The proposed analytical methodology is an improvement that allows more accurate modeling of core performance. The NRC has reviewed and approved this methodology for use in lieu of the current methodology; thus, the margin of safety is not reduced due to this change.

Therefore, it is concluded that the proposed change does not involve a significant reduction in a margin of safety.

The NRC staff has reviewed the licensee's analysis and, based on this review, it appears that the three standards of 10 CFR 50.92(c) are satisfied. Therefore, the NRC staff proposes to determine that the amendment request involves no significant hazards consideration.

Attorney for licensee: Jay Silberg, Esq., Pillsbury Winthrop Shaw Pittman LLP, 2300 N Street, NW, Washington, DC 20037.

NRC Branch Chief: Michael T. Markley.

Notice of Issuance of Amendments to Facility Operating Licenses and Combined Licenses

During the period since publication of the last biweekly notice, the Commission has issued the following amendments. The Commission has determined for each of these amendments that the application complies with the standards and requirements of the Atomic Energy Act of 1954, as amended (the Act), and the Commission's rules and regulations. The Commission has made appropriate findings as required by the Act and the Commission's rules and regulations in 10 CFR Chapter I, which are set forth in the license amendment.

A notice of consideration of issuance of amendment to facility operating license or combined license, as applicable, proposed no significant hazards consideration determination, and opportunity for a hearing in connection with these actions, was published in the *Federal Register* as indicated.

Unless otherwise indicated, the Commission has determined that these amendments satisfy the criteria for categorical exclusion in accordance with 10 CFR 51.22. Therefore, pursuant to 10 CFR 51.22(b), no environmental impact statement or environmental assessment need be prepared for these amendments. If the Commission has prepared an environmental assessment under the special circumstances provision in 10 CFR 51.22(b) and has made a determination based on that assessment, it is so indicated.

For further details with respect to the action see (1) the applications for amendment, (2) the amendment, and (3) the Commission's related letter, Safety Evaluation and/or Environmental Assessment as indicated. All of these items are available for public inspection at the Commission's Public Document Room (PDR), located at One White Flint North, Room O1-F21, 11555 Rockville Pike (first floor), Rockville, Maryland 20852. Publicly available documents created or received at the NRC are accessible electronically through the Agencywide Documents Access and Management System (ADAMS) in the NRC Library at http://www.nrc.gov/reading-rm/adams.html. If you do not have access to ADAMS or if there are problems in accessing the documents located in ADAMS, contact the PDR's Reference staff at 1-800-397-4209, 301-415-4737 or by email to pdr.resource@nrc.gov.

Arizona Public Service Company, et al., Docket Nos. STN 50-528, STN 50-529; and STN 50-530, Palo Verde Nuclear Generating Station, Units 1, 2, and 3, Maricopa County, Arizona

Date of application for amendment: December 12, 2012.

Brief description of amendment: The amendments revised the Technical Specifications (TSs) relating to reactor coolant system (RCS) activity limits by replacing the current TS limits on primary coolant gross specific activity with limits on primary coolant noble gas activity. The noble gas activity would reflect a new DOSE EQUIVALENT XE-133 definition that would replace the current E-bar average disintegration energy definition. The changes are consistent with NRC-approved Industry/Technical Specifications Task Force (TSTF) Standard Technical Specification change traveler, TSTF-490, Revision 0, "Deletion of E-bar Definition and Revision to RCS [Reactor Coolant System] Specific Activity Technical Specifications," with deviations.

Date of issuance: November 25, 2013.

Effective date: As of the date of issuance and shall be implemented within 180 days from the date of issuance.

Amendment No.: Unit 1 - 192; Unit 2 - 192; Unit 3 - 192.

Renewed Facility Operating License Nos. NPF-41, NPF-51; and NPF-74: The amendment revised the Operating Licenses and Technical Specifications.

Date of initial notice in *Federal Register*: March 4, 2013 (78 FR 14128).

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 25, 2013.

No significant hazards consideration comments received: No.

Dominion Nuclear Connecticut, Inc., Docket No. 50-336, Millstone Power Station, Unit 2, New London County, Connecticut

Date of amendment request: April 3, 2013.

Description of amendment request: The amendment would revise Technical Specification 3.9.16 "Shielded Cask," due to changes to the minimum decay time for fuel assemblies adjacent to the spent fuel pool cask laydown area.

Date of issuance: November 14, 2013.

Effective date: As of the date of issuance, and shall be implemented within 30 days.

Amendment No.: 316.

Renewed Facility Operating License No. DPR-65: Amendment revised the License and Technical Specifications.

Date of initial notice in *Federal Register*: June 11, 2013 (78 FR 35062).

The Commission's related evaluation of the amendment is contained in a Safety Evaluation dated November 14, 2013.

No significant hazards consideration comments received: No.

PSEG Nuclear LLC, Docket Nos. 50-272 and 50-311, Salem Nuclear Generating Station, Units 1 and 2, Salem County, New Jersey

Date of amendment requests: November 30, 2012, as supplemented by letter dated May 31, 2013.

Brief description of amendments: The amendments approve a change to the site Emergency Plan to remove the backup plant vent extended range noble gas radiation monitoring (R45) indication, recording, and alarm capability in the emergency response facilities. Although the R45B/C monitor equipment skid will be removed, the licensee will maintain a capability in its

Emergency Plan to take post-accident samples from the plant vent stack, as specified by an earlier commitment to Regulatory Guide 1.97, "Instrumentation for Light-Water-Cooled Nuclear Power Plants to Assess Plant and Environs Conditions During and Following an Accident."

Date of issuance: November 27, 2013.

Effective date: As of the date of issuance and shall be implemented within 60 days.

Amendment Nos.: 305 and 287.

Renewed Facility Operating License Nos. DPR-70 and DPR-75: The amendments revised the Facility Operating License and approved revisions to the Emergency Plan.

Date of initial notice in *Federal Register*: May 14, 2013 (78 FR 28252). The supplemental letter dated May 31, 2013, provided information that clarified the application, did not expand the scope of the application as originally noticed, and did not change the staff's original proposed no significant hazards consideration determination as published in the *Federal Register*.

The Commission's related evaluation of the amendments is contained in a Safety Evaluation dated November 27, 2013.

No significant hazards consideration comments received: No.

Dated at Rockville, Maryland, this 2nd day of December 2013.

For the Nuclear Regulatory Commission.

Michele G. Evans, Director,
Division of Operating Reactor Licensing,
Office of Nuclear Reactor Regulation.

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